PAKS NUCLEAR POWER PLANT LTD.
Development and Analysis Section
Person: Éva Tóth
☎: +36-75-508-986
e-mail: <tothnel@npp.hu>

Fifth Internatioanl Information Exchange Forum on Safety Analysis for NPP of VVER and RBMK Types
16-20 October, 2000, Obninsk, Russia

SAFETY ANALYSES SUPPORTING THE SYMPTOM ORIENTED EMERGENCY OPERATING PROCEDURES

Abstract:

New Emergency Operating Procedures were made for the Paks NPP by the leading of Westinghouse Electric Co. The Strategy Report of the optimal and functional recovery guidelines were made in the first stage of the work. During preparation of the Strategy Report safety analyses were defined the completion of which are essential for making the developed strategy definitive and for establishment of it. These analyses were carried out by the Atomic Energy Research Institute (KFKI-AEKI).

In this paper 11 main topics are specified (list of prepared analyses in particular see in Appendix 4) which were essential for the development and working out of procedures according to the Strategic Report. During the specification of analyses the concrete guidelines supported by the presented analysis were always given; the task was also interpreted and those preliminary considerations and assumption were introduced which were used for making analyses.

2. General Organisation of the EOPs

The EOPs are dealing with those events that actuate or require reactor trip or ESFAS operation. These events can further be categorised based on their features such as:

- events could be diagnosed based on symptoms (i.e. plant parameters)
- events could readily not amenable to diagnose on the basis of assumptions.

Corresponding to these two different types of emergency situations there are also two different types of procedures: event-dependent and event-independent part (see Appendix 1).

2.1. Event-dependent part

They are those procedures addressing emergency situations that can be readily diagnosed. These procedures aim at the optimal recovery for the plant for specifically diagnosed events: Optimal Recovery Guidelines (ORGs).

The purpose of Optimal Recovery Guidelines is to bring the plant to the optimal end-status for all events covered. The optimal recovery concept is based on the premise that radiation release and equipment damage can both minimised through associating the symptoms of an emergency transient with a predefined plant condition and implementing an associated predefined event-related recovery strategy to achieve an optimal end status. The optimal end state varies with emergency transient but is the state in which radiation release and equipment damage are minimised and plant conditions are stable with plant equipment operating in long term alignments.

The ORGs are entered when the reactor is tripped or the Emergency Core Cooling System is actuated. An immediate verification of the automatic protection signals is performed and the accident diagnosis process is initiated. When the nature of the accident is identified, the operator is directed to the applicable recovery procedure.

Three levels of diagnosis are built in the ORGs: early diagnosis (E-0), continuous and rediagnosis. Guidelines are provided to the adequate procedure if an error in diagnosis process is identified.

The ORGs are addressed to 4 major event categories (see Appendix 2):
- Loss of primary or secondary coolant
- Faulted steam generator
- Steam generator tube rupture
- Loss of all AC power

They provide recovery instructions for the „classical” accidents and for those cases when they are combined with the equipment failures or other unexpected events.

2.2. Event-independent part

They are those procedures addressing emergency situations that can’t be readily diagnosed. These procedures will be based on monitoring as well as restoring plant safety: Function Restoration Guidelines (FRGs)

a) Critical Safety Functions Monitoring

Early in the course of the accident, one has to initiate monitoring of the Critical Safety Functions. These CFSs are defined as a set of functions ensuring the integrity of the physical barriers against the releases of radioactive material:
- Fuel matrix/cladding
- Primary System Boundary
- Confinement Building

Monitoring of these functions is performed continuously through a cyclic application of the Status Trees:
- **S**ubcriticality
- **C**ore Cooling
- **H**eat Sink
- **P**rimary Integrity
- **C**onfinement Integrity
- **I**nventory

b) The Function Restoration Guidelines (FRGs)

The objective of the event-independent part of the Emergency Operating Procedures (EOPs) is to provide means to evaluate and restore the plant nuclear safety. The concept is based on the premise that radiation release to the environment can be minimised if the barriers to activity release are protected (barriers of defence in depth). In order to accomplish this goal, a set of functions has been defined which are critical from the plant nuclear safety point of view. These are the Critical Safety Functions. To be able to evaluate the status of these functions, Status Trees have been designed, one per CSF. Once the state of the CSF is evaluated, based on their state and the rules of priority one can designate a Function Restoration Guideline to be implemented for restoring CSF (see Appendix 3).

The Function Restoration Guidelines are entered when the Critical Safety Function monitoring identifies a challenge to one of the functions. Depending on the severity of the challenge, the transfer to a Function Restoration Guideline can be immediate for a severe challenge or delayed for a minor challenge. These guidelines are independent of the scenario of the accident, but only based on plant parameters and equipment availability.

3. Prepared analyses

There were carried out about 50 RELAP cases. In the following the results of required and prepared analyses and main conclusions are presented.

1. *Evaluation of the use of main loop isolation valves during accidents/ Isolation of primary leak by main loop isolation valves closure:*

   In the current Paks procedure dealing with primary break accident, credit is given to the closure of the main loop isolation valves. The following considerations apply:

   One of the fundamental basic rules of the EOPs is to not contradict nor violate the design bases of the plant. The loop isolation using the loop isolation valves has not been considered in the Paks accident analyses nor in the Final Safety Report, and as such, is not documented as a plant design basis. There were investigated some LOCA cases - medium and small sizes - with MLIV closure (see analysis No. 1. in Appendix 4). After MLIV closure on the broken loop RCS was filled up by 3 HPIS injection and there is less than 5 minutes to avoid the opening of PRZR relief valve. In conclusion, it was decided not to close MLIV if any of the HPIS pumps are in operation.

2. *Upper plenum coolability during natural circulation conditions:*

   Some analyses were made to determine the appropriate cooldown rate during natural circulation without steam void in vessel (see analysis No. 4. in Appendix 4). There were calculated NC cooldown with 20 °C/h and 14 °C/h rate. In conclusion, the 20 °C/h rate was too fast to assure RCS cooldown without void formation in the upper head, but 14 °C/h was found appropriate.
   The possible void formation in the SG primary collectors under natural circulation cooldown conditions was investigated. This phenomenon could be suppressed by using the degasifier lines.
   To calculate the boron concentration in the primary system is necessary to compensate boron acid for the non-boration of the "passive parts" (pressurizer, reactor vessel head, SG collectors). Optimum boron mixing were calculate in the analysis.
   Based on the transient analyses (see results of analysis No. 4. in Appendix 4) it was concluded that the 30 °C/h cooldown rate with 5 t/h upper head injection is appropriate to assure RCS cooldown without void formation in the upper head, but not in the steam generators. The Plant Engineering Staff should evaluated whether using EVS (and discharging the primary coolant into confinement) or waiting some hours is required.

3. *Malfunction of high pressure injection system at full power:*

   The specific scenario of a spurious start of HPIS needs particular attention. In such case, the three trains would start injecting because of modifications in the new reactor protection system. Due to the flow from the three HPIS pumps, the level will increase in

the pressurizer, thereby increasing the pressure, which is expected to result in actuation of the pressurizer spray. Based on the system characteristics (especially HPIS pumps capacity), there could be a potential for reaching the pressurizer solid condition (or at least spray nozzle flooding) and also of dead-heading for the HPIS pumps.
An analysis for a spurious HPIS actuation was made (see analysis No. 5. in Appendix 4). It was anyway be aimed at terminating the HPIS flow while protecting the RCS from the overpressure. Results show that primary pressure increasing fast caused by 3 HPIS injection and after 5 minutes PRZR relief valve opens (analysis was made with old-type-valve with opening setpoint 136 bar). After 15 minutes PRZR was filled up. In conclusion, on the units with old-type-valve the operator should intervene within 15 minutes. The new relief valves opening setpoint is 138 bar (this pressure is the end of HPIS injection) so the problem is eliminated.

4. *Cooldown without letdown during natural circulation:*

   Best estimate analysis could justify the strategy implemented in this procedure (see analysis No. 6. in Appendix 4) In conclusion, the primary boron concentration during the 14 °C/h natural circulation cooldown without letdown, is sufficient to keep the core in safe shutdown conditions.

5. *Review of criteria for safety injection termination:*

   Guideline ES-1.1 safety injection termination criteria do not verify boron concentration. Paks has verified that safety injection termination criteria need not be modified to include boron concentration verification (see analysis No. 6. in Appendix 4). From these results it can be concluded that in controlled cooldown cases, when the PRZR level is maintained by make-up pump injecting coolant with 40 g/kg boron concentration, there is no danger of recriticality and safe shutdown core conditions are provided.
   There was investigated another - more fast cooldown - streamline break transient (see analysis No. 8. in Appendix 4) whether subcriticality can be maintained also in this case. The calculation results indicate that HPIS injection could be terminated and there is no danger of recriticality. Due to the high amount of boron injected by 3 HPIS pumps the primary boron concentration increases very fast ensuring high margin when HPIS pumps can be stopped by the operator. Here should be noted, that there is no analysis for only the 1 HPIS injection (conservative accident analysis).

6. *Total loss of power:*

   There were investigated cases with or without cooldown by secondary side and with or without loss of primary cooling 5 t/h flow on each 6 MCP sealing (see analysis No. 10 and 11. in Appendix 4). In the cases without primary leakage the time available for recovery - even with secondary depressurisation - exceeds 10 hours. This time reduces to less than 5 hours without secondary bleed, if the pump seals are defected. In conclusion, the secondary bleed should be initiated if only a loss of primary coolant also exists. Otherwise the secondary side depressurisation can lead to shorter available time for recovery.

7. *Main steam header rupture:*

   The existing analysis for the uncontrolled depressurisation of all steam generators is based on a scenario where neider main steam header separation nor steam generator isolation occurs, enabling all steam generators to blow down through the ruptured main steam line (i.e. the isolation valve on the ruptured line fails to close), however the faulted SG is automatically isolated on feed side by the logic. For this scenario, the analysis was finalised to include safety injection termination criterion for boron concentration.

Compensation of shrinkage was investigated in the analysis No. 8. (see Appendix 4) to determine the time of possible feedwater de-isolation by the operator, when the RCS cooldown rate falls below the 60 °C/h limit value. This occurs at 19 minutes when the RCPs are running and at 26 minutes when the MCPs are stopped. These time periods are sufficiently long to permit the operator to intervene and de-isolate feedwater flow according to this procedure. However, it should be noted that the normal feedwater reserves will be exhausted by about 25 minutes.

8. *Pressure reduction by opening of pressurizer safety valves during hot leg/cold leg SBLOCA:*

The opening of the pressurizer safety valves or RCS pressure control has the lowest priority in the ERG. The effectiveness/benefit of opening the safety valves depends on its size. Paks specific analysis of the effectiveness of the safety valves was provided (see analysis No. 9. in Appendix 4). There were calculated cases with or without HA injection, with operator actions opening of PRZR safety valves and initiation secondary side depressurisation in FR-C.2. The results show that LPIS injection can be initiated temporarily by the primary bleed and core damage time could be delayed. However without early (starting in FR-C.2) secondary bleed this action could not provide long term LPIS flow because of increasing primary pressure by steam formation in the uncovered core. If the secondary side depressurisation is initiated only in FR-C.1 procedure that may occur faster reaching of 1200 °C safety limit of cladding temperature.

There are some differences in that case when HA injection is available. Results show (see analysis No. 13. in Appendix 4) that in this case starting secondary bleed in FR-C.2 (at 370 °C) is sufficient and stable LPIS injection could be provided.
Calculations show that, if HPIS injection is not available, early start of secondary bleed is very important to avoid core damage in FR-C.1, especially in case without HA injection. If so, the secondary side depressurisation initiated at 370 °C core outlet temperature can avoid extended core damage, although short term violation of cladding temperature limit could occur.

9. *Total loss of power with primary feed and bleed, analysis of the optimum B&F configuration:*

The following Paks-specific analyses and evaluations were required (see analysis No. 10. in Appendix 4)

- determine long term low pressure feed sources to SGs, and the capability to cool and depressurise the system to these low pressure sources with the SG inventory;
- based on the above analysis, determine the entry point of FR-H.1; it was decided to use symptoms: total feedwater flow < 65 t/h and all SG wide range level < 700 mm.
- determine the transition point to primary Bleed and Feed (based on B&F analysis).

Based on analyses, the operator should initiate secondary bleed if there is no chance to restore any type of feedwater more than 8 hours.

There were 3 cases calculated using results of calculations of total loss of power. Primary B&F was initiated at different core outlet temperatures (320 °C, 500 °C and 650 °C). Results show that bleed and feed process was effective in all 3 cases. The overall conclusion that operator can wait until core outlet temperature begin to rise and primary B&F should initiate at 300 mm in all SGs.
The optimal configuration for the B&F process was determined:
- 1 HPIS pump,
- 1 new PRZR relief valve (with 50 t/h capacity) or 1 PRZR safety valve,
- without HA injection.

The arguments are in following:
- even 1 HPIS cold water injection leads to violation of the 60 °C/h limit and is enough to borate the RCS and prevent core damage,
- with both configuration the process is successful,
- HA operation decreases the temperature quickly and fills up the PRZR early.

10. *P-T diagram determination.*

The strategy given in the generic guideline should be applicable to Paks NPP, but specific analysis has been performed to establish the soak time requirements (see analysis No. 12. in Appendix 4)

11. *Post LOCA cool-down:*

There were made some LOCA calculations with 33 mm breaksize - when the break flow can be compensated by 3 HPIS pumps at a primary pressure around 100 bar (see analysis No. 12. in Appendix 4) to examine the effectiveness of the major strategy of ES-1.2 procedure. It was checked whether sufficient subcooling and subcriticality can be assured throughout the coooldown process. Finally it was determined, that termination of the first HPIS pump is successful. But after the termination of the second HPIS pump it was restarted by the ECCS signal on low PRZR level. In conclusion, this breaksize is allowed to reduce HPIS injection but not to terminate all of 3 HPIS pumps.

Conclusions:

All of the calculations were carried out by RELAP5/MOD 3.2 by KFKI-AEKI and on the basis of results most of questions have been answered. Consequences were included into the Paks-specific guidelines.

References:

[1] Emergency operation guideline strategies for NPP Paks, Barnabás Bognár, Paks, May 2000.

[2] Isolation of primary leak by main loop isolation valves closure, Allorsz/R01/A, KFKI-AEKI, Budapest, May 1998.

[3] Summary of the PRISE project with respect to application of MGVs, Allorsz/R01/B, KFKI-AEKI, Budapest, September 1998.

[4] Evaluation of the use of primary loop isolation valves during accidents, Allorsz/R01/C, KFKI-AEKI, Budapest, January 1999.

[5] Upper plenum coolability during natural circulation conditions, Allorsz/R02/A, KFKI-AEKI, Budapest, December 1998.

[6] Steam generator collector head coolability during natural circulation conditions, Allorokt/R02/A, Budapest, February 1999.

[7] Malfunction of high pressure injection system at full power, Allorsz/R03/A, KFKI-AEKI, Budapest, August 1998.

[8] Cooldown without letdown during natural circulation including the boration aspect of SI termination, Allorsz/R04/A, KFKI-AEKI, Budapest, September 1998.

[9] Total loss of power, Allorsz/R06/A, KFKI-AEKI, Budapest, September/December 1998.

[10] Main steam header rupture, Allorsz/R07/A, KFKI-AEKI, Budapest, August/December 1998.

[11] Pressure reduction by opening of pressuriser safety valves during hot leg SB-LOCA, Allorsz/R08/A, KFKI-AEKI, Budapest, December 1998.

[12] Total loss of power with primary feed and bleed, Allorsz/R09/A, KFKI-AEKI, Budapest, September/December 1998.

[13] Total loss of power with primary feed and bleed, analysis of the optimum B&F configuration, Allorsz/R09/B, KFKI-AEKI, Budapest, March 1999.

[14] Pressure reduction by opening of pressurizer safety valves during cold leg SB-LOCA, Allorokt/R01/A, KFKI-AEKI, Budapest, June 1999.

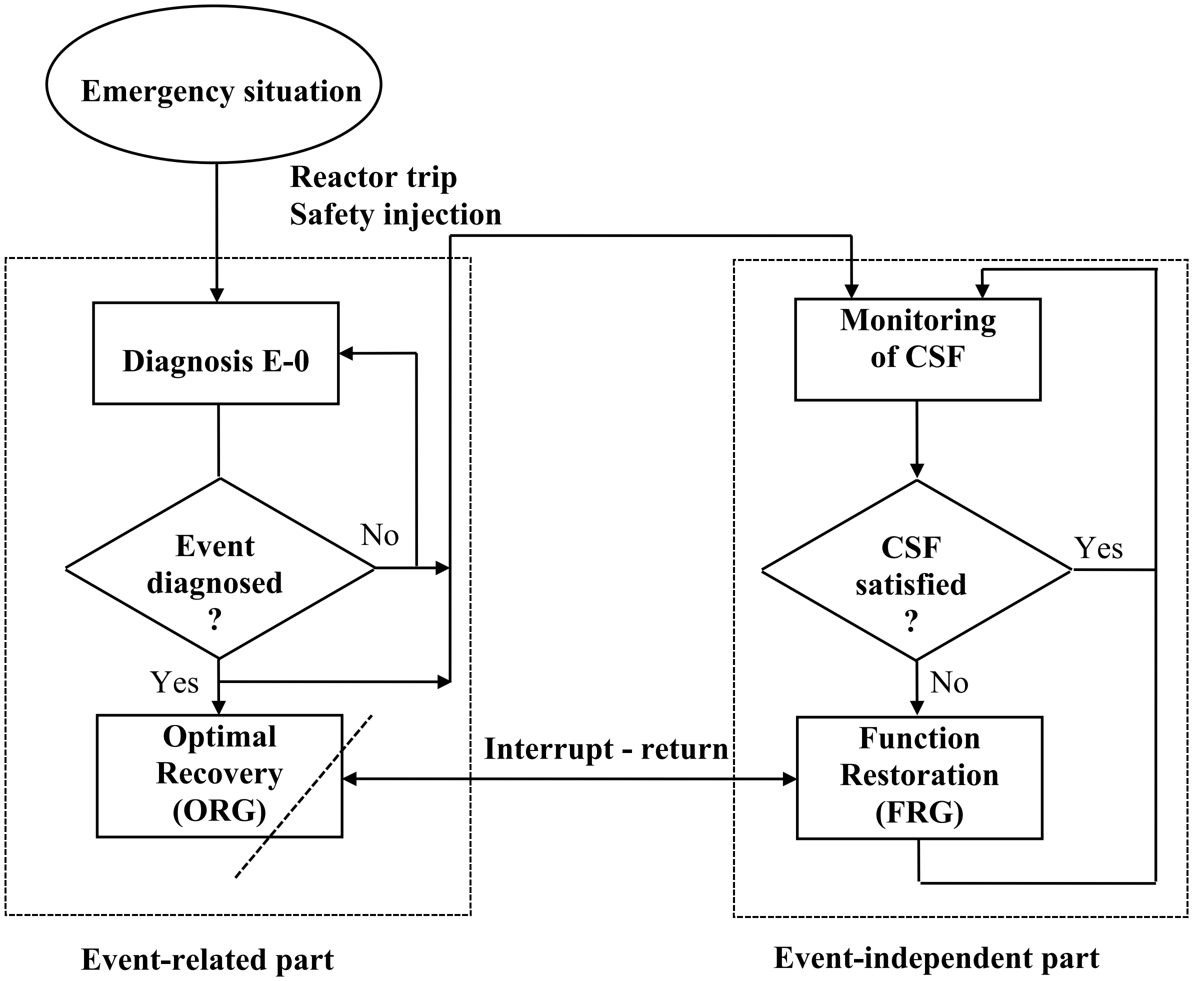
[15] Specification of pressure-temperature limit-curves, Allorsz/R10/A, September 1998, KFKI-AEKI, Budapest,

[16] Post LOCA cooldown, Allorsz/R11/A, December 1998, Allorsz/R11/B, September 1999, KFKI-AEKI, Budapest.

[17] PHARE 4.2.7.a/93 Project: Beyond Design Basis Accident Analysis and Acident Management, Accident Management Study, Westinghouse Energy Systems Europe S.A., March 1998

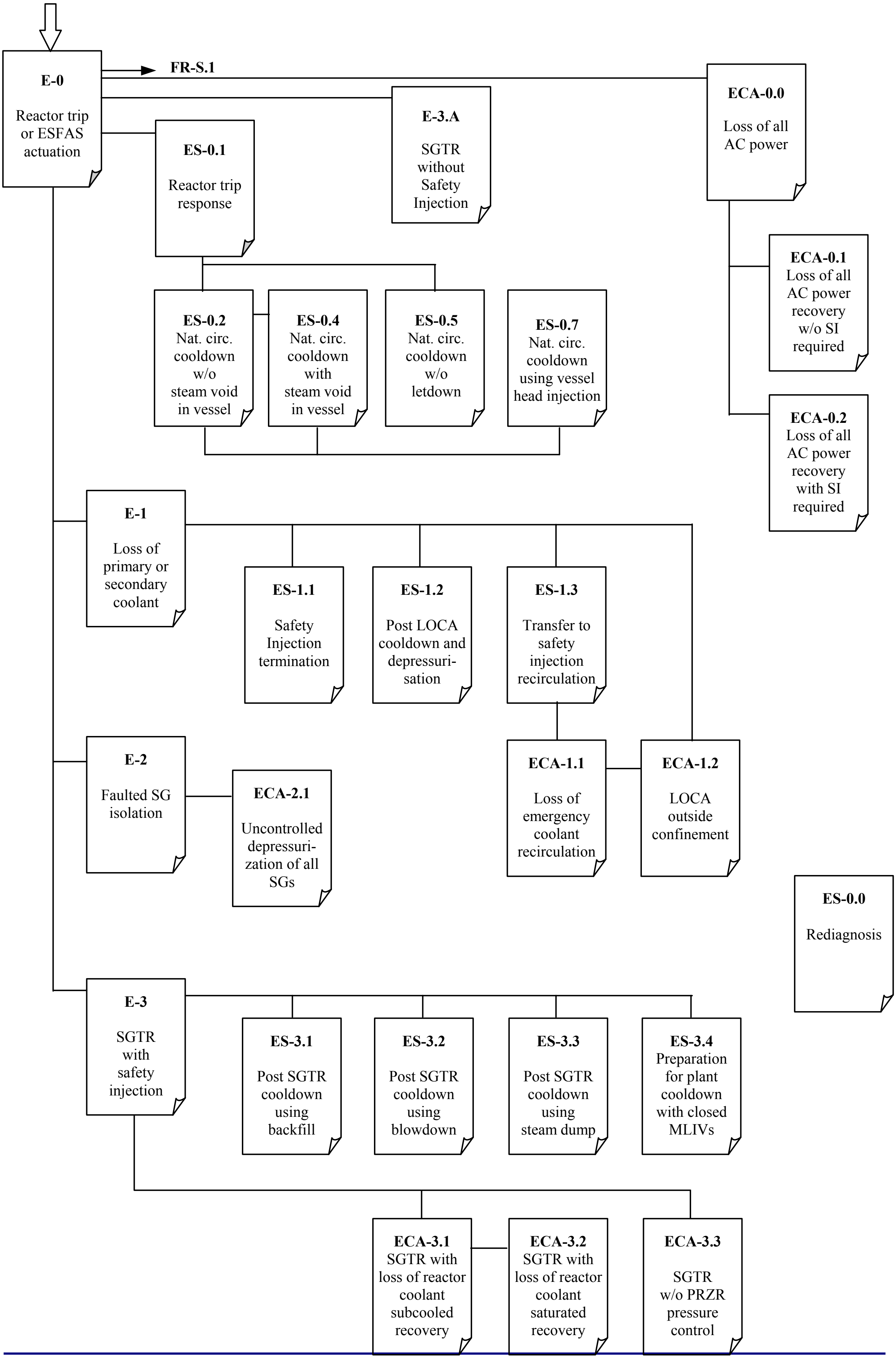
APPENDIX 1

General Organization of the EOPs



APPENDIX 2

ORGs structure



APPENDIX 3

FRGs structure

F-0.1 Subcriticality	F-0.2 Core Cooling	F-0.3 Heat Sink	F-0.4 Primary Integrity	F-0.5 Confinement Integrity	F-0.6 Inventory
FR-S.1 Response to nuclear power generation - ATWS	**FR-C.1** Response to inadequate core cooling	**FR-H.1** Response to loss of secondary heat sink	**FR-P.1** Response to imminent PTS condition	**FR-Z.1** Response to high confinement pressure	**FR-I.1** Response to high PRZR level
FR-S.2 Response to loss of core shutdown	**FR-C.2** Response to degreaded core cooling	**FR-H.2** Response to steam generator overpressur	**FR-P.2** Response to anticipated PTS	**FR-Z.2** Response to confinement flooding	**FR-I.2** Response to low PRZR level
	FR-C.3 Response to saturated core cooling	**FR-H.3** Response to steam generator high level		**FR-Z.3** Response to high confinement radiation	**FR-I.3** Response to voids in the reactor vessel
		FR-H.4 Response to loss of norm. steam release capability		**FR-Z.4** Response to low confinement pressure	
		FR-H.5 Response to steam generator low level			

APPENDIX 4

List of Analyses and Specific Studies

1.	Allorsz/R01/A	Isolation of primary leak by main loop isolation valves closure.
2.	Allorsz/R01/B	Summary of the PRISE project with respect to applicationof MGVS
3.	Allorsz/R01/C	Evaluation of the use of primary loop isolation valves during accidents.
4.	Allorsz/R02/A	Upper plenum coolability during natural circulation conditions
5.	Allorsz/R03/A	Malfunction of high pressure injection system at full power.
6.	Allorsz/R04/A	Cooldown without letdown during natural circulation including the boration aspect of SI termination.
7.	Allorsz/R06/A	Total loss of power.
8.	Allorsz/R07/A	Main steam header rupture.
9.	Allorsz/R08/A	Pressure reduction by opening of pressureser safety valves during hot leg SBLOCA
10.	Allorsz/R09/A	Total loss of power with primary feed and bleed
11.	Allorsz/R09/B	Total loss of power with primary feed and bleed, analysis of the optimum B&F configuration.
12.	Allorsz/R10/A	P-T diagram determination.
13.	Allorsz/R11/B	Post LOCA cool-down.
14.	Allorokt/R01/A	Pressure reduction by opening of pressuriser safety valves during cold leg SBLOCA.
15.	Allorokt/R02	Steam Generator collector head coolability during natural circulation conditions.